UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DEC 15 1983

Docket No. 50-416

MEMORANDUM FOR: Cecil O. Thomas, Chief
Standardization & Special Projects Branch
Division of Licensing

FROM: B. D. Liaw, Chief
Materials Engineering Branch
Division of Engineering

SUBJECT: PROOF AND REVIEW OF GRAND GULF UNIT 1
TECHNICAL SPECIFICATIONS

As requested by your letter dated October 4, 1983, the Materials Engineering Branch, Division of Engineering, has reviewed the Grand Gulf Unit 1 Technical Specifications. The sections reviewed were 4.0.5, B4.0.5, and 3/4.4.6 on pages 3/4.4-17 thru 3/4.4-20. These sections are acceptable, except that the pressure-temperature limit curves in Figure 3.4.6.1-1 do not comply with the closure flange pressure temperature safety margins in Paragraph IV.A.2 of Appendix G, 10 CFR 50, which became effective on July 26, 1983. We will be sending to all licensees/applicants a generic letter informing them of the revised regulatory requirements. In accordance with that letter, the pressure-temperature limit curves in Figure 3.4.6.1-1 may require revision.

This comment has been coordinated with Mr. Donald R. Hoffman of your staff.

B. D. Liaw, Chief
Materials Engineering Branch
Division of Engineering

cc: R. Vollmer
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Appendix G—Fracture Toughness Requirements

Table of Contents

I. Introduction and Scope

This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

The ASME Code forms the basis for the requirements of this Appendix. "ASME Code" means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. If no section is specified, the reference is to Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components." "Section XI" means Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components." If no edition or addenda is specified the applicable ASME Code edition and addenda and any limitations and modifications thereof are specified in § 50.55a of this part.

The ASME Boiler and Pressure Vessel Code has been approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. It is also available for inspection at the Nuclear Regulatory Commission's Public Document Room, 1717 H Street NW., Washington, D.C.

The requirements of this appendix apply to the following materials:

NOTE.—The adequacy of the fracture toughness of other ferritic materials not covered in this section shall be demonstrated to the Director, Office of Nuclear Reactor Regulation, on an individual case basis.

A. Carbon and low-alloy ferritic steel plate, forgings, castings, and pipe with specified minimum yield strengths not over 50,000 psi (345 MPa), and to those with specified minimum yield strengths greater than 50,000 psi (345 MPa) but not over 90,000 psi (621 MPa) if qualified by using methods equivalent to those described in paragraph G–2110 of the ASME Code as defined in paragraph II.A. of this appendix. The latest edition and addenda permitted by paragraph 50.55a(b) of this part at the time the analysis is made is to be used for the purpose of this paragraph.

B. Welds and weld heat-affected zones in the materials specified in paragraph I.A. of this appendix.

C. Materials for bolting and other types of fasteners with specified minimum yield strengths not over 130,000 psi (896 MPa).

48 FR 24008

II. Definitions

A. "Ferritic material" means carbon and low-alloy steels, higher alloy steels including all stainless alloys of the 4xx series, and maraging and precipitation hardening steels with a predominantly body-centered cubic crystal structure.

B. "System hydrostatic tests" means all preoperational system leakage and hydrostatic pressure tests and all system leakage and hydrostatic pressure tests performed during the service life of the pressure boundary in compliance with the ASME Code, Section XI.

C. "Specified minimum yield strength" means the minimum yield strength (in the unirradiated condition) of a material specified in the construction code under which the component is built under § 50.55a of this part.

D. "Reference temperature" means the reference temperature, RT_{NDT}, as defined in the ASME Code.

E. "Adjusted reference temperature" means the reference temperature as adjusted for irradiation effects (see Section V of this Appendix) by adding to RT_{NDT} the temperature shift, measured at the 30 ft-lb (41J) level, in the average Charpy curve for the irradiated material relative to that for the unirradiated material.

F. "Beltline" or "Beltline region of reactor vessel" means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

III. Fracture Toughness Tests

A. To demonstrate compliance with the fracture toughness requirements of Sections IV and V of this appendix, ferritic materials must be tested in accordance with the ASME Code and, for the beltline materials, the test requirements of Appendix H of this part. For a reactor vessel that was constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition (under § 50.55a of this part), the fracture toughness data and data analyses must be supplemented in a manner approved by the Director, Office of Nuclear Reactor Regulation, to demonstrate equivalence with the fracture toughness requirements of this Appendix.

B. Test methods for supplemental fracture toughness tests described in paragraph V.C.2. of this appendix must be submitted to and approved by the Director, Office of Nuclear Reactor Regulation, prior to testing.

C. All fracture toughness test programs conducted in accordance with paragraphs A and B of this section must comply with ASME Code requirements for calibration of test equipment, qualification of test personnel, and retention of records of these functions and of the test data.

IV. Fracture Toughness Requirements

A. The pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials must meet the requirements of the ASME Code supplemented as follows for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences:

1. Reactor vessel beltline materials must have Charpy upper-shelf energy [1] of no less than 75 ft-lb (102J) initially and must maintain upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. The latest edition and addenda of the ASME Code permitted by paragraph 50.55a(b) of this part at the time the analysis is made are to be used for the purposes of paragraphs IV.A.1 and IV.A.2 of this appendix.

2. When the core is not critical, pressure-temperature limits for the reactor vessel must be at least as conservative as those obtained by following the methods of analysis and the required margins of safety of Appendix G of the ASME Code supplemented by the requirements of Section V of this appendix. In addition, when pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions that are highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F (67°C) for normal operation and by 90°F (50°C) for hydrostatic pressure tests and leak tests, unless a lower temperature can be justified by showing that the margins of safety for those regions when they are controlling are equivalent to those required for the beltline when it is controlling. The justification submitted for the pressure temperature limits must describe the methods of analysis used.

3. When the core is critical (other than for the purpose of low-level physics tests), the temperature of the reactor vessel must not be lower than 40°F (22°C) above the minimum permissible temperature of paragraph 2. of this section nor lower than the minimum permissible temperature for the inservice system hydrostatic pressure test. An exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the preservice system hydrostatic test pressure. In this case the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload.

4. If there is no fuel in the reactor during system hydrostatic pressure tests or leak tests, the minimum permissible test temperature must be 60°F (33°C) above the adjusted reference temperature of the reactor vessel material in the region that is controlling (as specified in paragraph IV.A.2 of this appendix).

5. If there is fuel in the reactor during system hydrostatic pressure tests or leak tests, the requirements of paragraphs 2 or 3 of this section apply, depending on whether the core is critical during the test.

B. Reactor vessels for which the predicted

[1] Defined in ASTM E 185–79 and –82 which are incorporated by reference in Appendix H.

value of upper-shelf energy at end of life is below 50 ft-lb or for which the predicted value of adjusted reference temperature at end of life exceeds 200°F (93°C) must be designed to permit a thermal annealing treatment at a sufficiently high temperature to recover material toughness properties of ferritic materials of the reactor vessel beltline.

V. Inservice Requirements—Reactor Vessel Beltline Material

A. The effects of neutron radiation on the reference temperature and upper shelf energy of reactor vessel beltline materials, including welds, are to be predicted from the results of pertinent radiation effects studies in addition to the results of the surveillance program of Appendix H to this part.

B. Reactor vessels may continue to be operated only for that service period within which the requirements of Section IV of this appendix are satisfied using the predicted value of the adjusted reference temperature and the predicted value of the upper-shelf energy at the end of the service period to account for the effects of radiation on the fracture toughness of the beltline materials. These predictions are to be made for the radiation conditions at the critical location on the crack front of the assumed flaw.[3] The highest adjusted reference temperature and the lowest upper-shelf energy level of all the beltline materials must be used to verify that
48 FR 24008
the fracture toughness requirements are satisfied.

C. In the event that the requirements of Section V.B. of this appendix cannot be satisfied, reactor vessels may continue to be operated provided all of the following requirements are satisfied:

1. A volumetric examination of 100 percent of the beltline materials that do not satisfy the requirements of Section V.B. of this appendix is made and any flaws characterized according to Section XI of the ASME Code and as otherwise specified by the Director, Office of Nuclear Reactor Regulation.

2. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation is to be obtained from results of supplemental fracture toughness tests.

3. An analysis is performed that conservatively demonstrates, making appropriate allowances for all uncertainties, the existence of equivalent margins of safety for continued operation.

D. If the procedures of Section V.C. of this appendix do not indicate the existence of an equivalent safety margin, the reactor vessel beltline may, subject to the approval of the Director, Office of Nuclear Reactor Regulation, be given a thermal annealing treatment to recover the fracture toughness of the material. The degree of recovery is to be measured by testing additional specimens that have been withdrawn from the surveillance program capsules and that have been annealed under the same time-at-temperature conditions as those given the beltline material. The results, together with the results of other pertinent annealing-effects studies, are to provide the basis for establishing the adjusted reference temperature and upper-shelf energy after annealing. The reactor vessel may continue to be operated only for that service period within which the predicted fracture toughness of the beltline region materials satisfies the requirements of Section IV.A. of this appendix using the values of adjusted reference temperature and upper-shelf energy that include the effects of annealing and subsequent irradiation.

E. The proposed programs for satisfying the requirements of Sections V.C. and V.D. of this appendix are to be reported to the Director, Office of Nuclear Reactor Regulation, as specified in § 50.4(a) of this Part, for review and approval on an individual case basis at least 3 years prior to the date when the predicted fracture toughness levels will no longer satisfy the requirements of section V B of this appendix.

[3] For example, in analyses that follow Appendix G of the ASME Code, the radiation conditions to be used are those predicted for the material one fourth of the way through the vessel wall, i.e., at the deepest point on the crack front of the postulated defect.

Appendix H—Reactor Vessel Material Surveillance Program Requirements

Table of Contents

I. Introduction

The purpose of the material surveillance program required by this Appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. These data will be used as described in Sections IV and V of Appendix G to this part.

ASTM E 185–73, –79 and –82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," which are referenced in the following paragraphs, have been approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of ASTM E 185–73, –79, and –82, may be obtained from the American Society for Testing and Materials, 1916 Race St., Philadelphia, PA 19103. Copies will be available for inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C.

II. Surveillance Program Criteria

A. No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence (E>1MEV) at the end of the design life of the vessel will not exceed 10^{17} n/cm².

B. Reactor vessels that do not meet the conditions of paragraph II.A. of this Appendix must have their beltline materials monitored by this Appendix.

1. That part of the surveillance program conducted prior to the first capsule withdrawal must meet the requirements of the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased. Later editions of ASTM E 185 may be used, but including only those editions through 1982. For each capsule withdrawal after July 26, 1983, the test procedures and reporting requirements must meet the requirements of ASTM E 185–82 to the extent practical for the configuration of the specimens in the capsule. For each capsule withdrawal prior to July 26, 1983 either the 1973, the 1979, or the 1982 edition of ASTM E 185 may be used.

2. Surveillance specimen capsules must be located near the inside vessel wall in the beltline region so that the specimen irradiation history duplicates, to the extent practicable within the physical constraints of the system, the neutron spectrum, temperature history, and maximum neutron fluence experienced by the reactor vessel inner surface. If the capsule holders are attached to the vessel wall or to the vessel cladding, construction and inservice inspection of the attachments and attachment welds must be done according to the requirements for permanent structural
48 FR 24008
attachments to reactor vessels given in Sections III and XI of the ASME Code. The design and location of the capsule holders shall permit insertion of replacement capsules. Accelerated irradiation capsule may be used in addition to the required number of surveillance capsules specified in ASTM E 185.

3. A proposed withdrawal schedule must be submitted with a technical justification therefor to the Director, Office of Nuclear Reactor Regulation, for approval. The proposed schedule must be approved prior to implementation.

C. An integrated surveillance program may be considered for a set of reactors that have similar design and operating features. The representative materials chosen for surveillance from each reactor in the set may be irradiated in one or more of the reactors, but there must be an adequate dosimetry program for each reactor. No reduction in the requirements for number of materials to be irradiated, specimen types, or number of specimens per reactor is permitted, but the amount of testing may be reduced if the initial results agree with predictions. Integrated surveillance programs must be approved by the Director, Office of Nuclear Reactor Regulation, on a case-by-case basis. Criteria for approval include the following considerations:

1. The design and operating features of the reactors in the set must be sufficiently similar to permit accurate comparisons of the predicted amount of radiation damage as a function of total power output.

2. There must be adequate arrangement for data sharing between plants.

3. There must be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

Item No: 219 TECHNICAL SPECIFICATION PROBLEM SHEET Priority 5

______ / ______
Identified By Date

______
Responsible Supervisor

Tech Spec Reference: Figure 3.4.6.1-1

Problem Title: PRESSURE/TEMPERATURE LIMIT CURVES

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other): THE PRESSURE-TEMPERATURE LIMIT CURVES DO NOT COMPLY WITH THE CLOSURE FLANGE PRESSURE-TEMPERATURE SAFETY MARGINS IN PARAGRAPH IV.A.2 OF APP. G TO 10 CFR 50.

PROOF AND REVIEW COMMENT FROM MEB, DECEMBER 15, 1983.

2. Safety Significance:

3. Anticipated Resolution: EVALUATE TO DETERMINE IF THE FIGURE WILL REQUIRE REVISION.

4. NRC Response to Item (NRR/IE):

NRC Notified: ______ ______ / ______
Individual Notified Date Time

5. Disposition:

Items Closed: (How)

______ / ______
Date Time

cc: J. E. Cross
R. F. Rogers

Appendix G—Fracture Toughness Requirements

Table of Contents

I. Introduction and Scope

This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

The ASME Code forms the basis for the requirements of this Appendix. "ASME Code" means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. If no section is specified, the reference is to Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components." "Section XI" means Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components." If no edition or addenda is specified the applicable ASME Code edition and addenda and any limitations and modifications thereof are specified in § 50.55a of this part.

48 FR 24008

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The requirements of this appendix apply to the following materials:

Note.—The adequacy of the fracture toughness of other ferritic materials not covered in this section shall be demonstrated to the Director, Office of Nuclear Reactor Regulation, on an individual case basis.

A. Carbon and low-alloy ferritic steel plate, forgings, castings, and pipe with specified minimum yield strengths not over 50,000 psi (345 MPa), and to those with specified minimum yield strengths greater than 50,000 psi (345 MPa) but not over 90,000 psi (621 MPa) if qualified by using methods equivalent to those described in paragraph G-2110 of the ASME Code as defined in paragraph II.A. of this appendix. The latest edition and addenda permitted by paragraph 50.55a(b) of this part at the time the analysis is made is to be used for the purpose of this paragraph.

B. Welds and weld heat-affected zones in the materials specified in paragraph I.A. of this appendix.

C. Materials for bolting and other types of fasteners with specified minimum yield strengths not over 130,000 psi (896 MPa).

II. Definitions

A. "Ferritic material" means carbon and low-alloy steels, higher alloy steels including all stainless alloys of the 4xx series, and maraging and precipitation hardening steels with a predominantly body-centered cubic crystal structure.

B. "System hydrostatic tests" means all preoperational system leakage and hydrostatic pressure tests and all system leakage and hydrostatic pressure tests performed during the service life of the pressure boundary in compliance with the ASME Code, Section XI.

C. "Specified minimum yield strength" means the minimum yield strength (in the unirradiated condition) of a material specified in the construction code under which the component is built under § 50.55a of this part.

D. "Reference temperature" means the reference temperature, RT_{NDT}, as defined in the ASME Code.

E. "Adjusted reference temperature" means the reference temperature as adjusted for irradiation effects (see Section V of this Appendix) by adding to RT_{NDT} the temperature shift, measured at the 30 ft-lb (41J) level, in the average Charpy curve for the irradiated material relative to that for the unirradiated material.

48 FR 24008

F. "Beltline" or "Beltline region of reactor vessel" means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

III. Fracture Toughness Tests

A. To demonstrate compliance with the fracture toughness requirements of Sections IV and V of this appendix, ferritic materials must be tested in accordance with the ASME Code and, for the beltline materials, the test requirements of Appendix H of this part. For a reactor vessel that was constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition (under § 50.55a of this part), the fracture toughness data and data analyses must be supplemented in a manner approved by the Director, Office of Nuclear Reactor Regulation, to demonstrate equivalence with the fracture toughness requirements of this Appendix.

B. Test methods for supplemental fracture toughness tests described in paragraph V.C.2. of this appendix must be submitted to and approved by the Director, Office of Nuclear Reactor Regulation, prior to testing.

C. All fracture toughness test programs conducted in accordance with paragraphs A and B of this section must comply with ASME Code requirements for calibration of test equipment, qualification of test personnel, and retention of records of these functions and of the test data.

IV. Fracture Toughness Requirements

A. The pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials must meet the requirements of the ASME Code supplemented as follows for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences:

1. Reactor vessel beltline materials must have Charpy upper-shelf energy [1] of no less than 75 ft-lb (102J) initially and must maintain upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. The latest edition and addenda of the ASME Code permitted by paragraph 50.55a(b) of this part at the time the analysis is made are to be used for the purposes of paragraphs IV.A.1 and IV.A.2 of this appendix.

2. When the core is not critical, pressure-temperature limits for the reactor vessel must be at least as conservative as those obtained by following the methods of analysis and the required margins of safety of Appendix G of the ASME Code supplemented by the requirements of Section V of this appendix. In addition, when pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions that are highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F (67°C) for normal operation and by 90°F (50°C) for hydrostatic pressure tests and leak tests, unless a lower temperature can be justified by showing that the margins of safety for those regions when they are controlling are equivalent to those required for the beltline when it is controlling. The justification submitted for the pressure temperature limits must describe the methods of analysis used.

48 FR 24008

3. When the core is critical (other than for the purpose of low-level physics tests), the temperature of the reactor vessel must not be lower than 40°F (22°C) above the minimum permissible temperature of paragraph 2. of this section nor lower than the minimum permissible temperature for the inservice system hydrostatic pressure test. An exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the preservice system hydrostatic test pressure. In this case the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload.

4. If there is no fuel in the reactor during system hydrostatic pressure tests or leak tests, the minimum permissible test temperature must be 60°F (33°C) above the adjusted reference temperature of the reactor vessel material in the region that is controlling (as specified in paragraph IV.A.2 of this appendix).

5. If there is fuel in the reactor during system hydrostatic pressure tests or leak tests, the requirements of paragraphs 2 or 3 of this section apply, depending on whether the core is critical during the test.

B. Reactor vessels for which the predicted

[1] Defined in ASTM E 185-79 and -82 which are incorporated by reference in Appendix H.

value of upper-shelf energy at end of life is below 50 ft-lb or for which the predicted value of adjusted reference temperature at end of life exceeds 200°F (93°C) must be designed to permit a thermal annealing treatment at a sufficiently high temperature to recover material toughness properties of ferritic materials of the reactor vessel beltline.

V. Inservice Requirements—Reactor Vessel Beltline Material

A. The effects of neutron radiation on the reference temperature and upper shelf energy of reactor vessel beltline materials, including welds, are to be predicted from the results of pertinent radiation effects studies in addition to the results of the surveillance program of Appendix H to this part.

B. Reactor vessels may continue to be operated only for that service period within which the requirements of Section IV of this appendix are satisfied using the predicted value of the adjusted reference temperature and the predicted value of the upper-shelf energy at the end of the service period to account for the effects of radiation on the fracture toughness of the beltline materials. These predictions are to be made for the radiation conditions at the critical location on the crack front of the assumed flaw.[2] The highest adjusted reference temperature and the lowest upper-shelf energy level of all the beltline materials must be used to verify that the fracture toughness requirements are satisfied.

C. In the event that the requirements of Section V.B. of this appendix cannot be satisfied, reactor vessels may continue to be operated provided all of the following requirements are satisfied:

1. A volumetric examination of 100 percent of the beltline materials that do not satisfy the requirements of Section V.B. of this appendix is made and any flaws characterized according to Section XI of the ASME Code and as otherwise specified by the Director, Office of Nuclear Reactor Regulation.

2. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation is to be obtained from results of supplemental fracture toughness tests.

3. An analysis is performed that conservatively demonstrates, making appropriate allowances for all uncertainties, the existence of equivalent margins of safety for continued operation.

D. If the procedures of Section V.C. of this appendix do not indicate the existence of an equivalent safety margin, the reactor vessel beltline may, subject to the approval of the Director, Office of Nuclear Reactor Regulation, be given a thermal annealing treatment to recover the fracture toughness of the material. The degree of recovery is to be measured by testing additional specimens that have been withdrawn from the surveillance program capsules and that have been annealed under the same time-at-temperature conditions as those given the beltline material. The results, together with the results of other pertinent annealing-effects studies, are to provide the basis for establishing the adjusted reference temperature and upper-shelf energy after annealing. The reactor vessel may continue to be operated only for that service period within which the predicted fracture toughness of the beltline region materials satisfies the requirements of Section IV.A. of this appendix using the values of adjusted reference temperature and upper-shelf energy that include the effects of annealing and subsequent irradiation.

E. The proposed programs for satisfying the requirements of Sections V.C. and V.D. of this appendix are to be reported to the Director, Office of Nuclear Reactor Regulation, as specified in § 50.4(a) of this Part, for review and approval on an individual case basis at least 3 years prior to the date when the predicted fracture toughness levels will no longer satisfy the requirements of section V B of this appendix.

[2] For example, in analyses that follow Appendix G of the ASME Code, the radiation conditions to be used are those predicted for the material one fourth of the way through the vessel wall, i.e., at the deepest point on the crack front of the postulated defect.

Appendix H—Reactor Vessel Material Surveillance Program Requirements

Table of Contents

I. Introduction

The purpose of the material surveillance program required by this Appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. These data will be used as described in Sections IV and V of Appendix G to this part.

ASTM E 185-73, -79 and -82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," which are referenced in the following paragraphs, have been approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of ASTM E 185-73, -79, and -82, may be obtained from the American Society for Testing and Materials, 1916 Race St., Philadelphia, PA 19103. Copies will be available for inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C.

II. Surveillance Program Criteria

A. No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence (E > 1MEV) at the end of the design life of the vessel will not exceed 10^{17} n/cm².

B. Reactor vessels that do not meet the conditions of paragraph II.A. of this Appendix must have their beltline materials monitored by this Appendix.

1. That part of the surveillance program conducted prior to the first capsule withdrawal must meet the requirements of the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased. Later editions of ASTM E 185 may be used, but including only those editions through 1982. For each capsule withdrawal after July 26, 1983, the test procedures and reporting requirements must meet the requirements of ASTM E 185-82 to the extent practical for the configuration of the specimens in the capsule. For each capsule withdrawal prior to July 26, 1983 either the 1973, the 1979, or the 1982 edition of ASTM E 185 may be used.

2. Surveillance specimen capsules must be located near the inside vessel wall in the beltline region so that the specimen irradiation history duplicates, to the extent practicable within the physical constraints of the system, the neutron spectrum, temperature history, and maximum neutron fluence experienced by the reactor vessel inner surface. If the capsule holders are attached to the vessel wall or to the vessel cladding, construction and inservice inspection of the attachments and attachment welds must be done according to the requirements for permanent structural attachments to reactor vessels given in Sections III and XI of the ASME Code. The design and location of the capsule holders shall permit insertion of replacement capsules. Accelerated irradiation capsule may be used in addition to the required number of surveillance capsules specified in ASTM E 185.

3. A proposed withdrawal schedule must be submitted with a technical justification therefor to the Director, Office of Nuclear Reactor Regulation, for approval. The proposed schedule must be approved prior to implementation.

C. An integrated surveillance program may be considered for a set of reactors that have similar design and operating features. The representative materials chosen for surveillance from each reactor in the set may be irradiated in one or more of the reactors, but there must be an adequate dosimetry program for each reactor. No reduction in the requirements for number of materials to be irradiated, specimen types, or number of specimens per reactor is permitted, but the amount of testing may be reduced if the initial results agree with predictions. Integrated surveillance programs must be approved by the Director, Office of Nuclear Reactor Regulation, on a case-by-case basis. Criteria for approval include the following considerations:

1. The design and operating features of the reactors in the set must be sufficiently similar to permit accurate comparisons of the predicted amount of radiation damage as a function of total power output.

2. There must be adequate arrangement for data sharing between plants.

3. There must be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

"TECH SPEC PRIORITY"

Punchlist Item # 219

Tech Spec Figure 3.4.6.1-1

Priority 1A

TO: Manager of Nuclear Plant Engineering

FROM: Chairman, Prioritization and Disposition Chairman

SUBJECT: Technical Specifications Punchlist Item # 219

PDTS:84/ 0011

DATE: 3/10/84

The subject Tech Spec item has been determined by the Disposition Committee to require Engineering support.

DETAILS: Evaluate the effect on Press/Temp Limit Curves (Fig. 3.4.6.1-1) of recent changes to 10CFR 50 App G, para IV.A.2.
Verify with GE that they are investigating this item. Follow up this item till resolved

Please contact Joe Hendry at Extension 267~~0~~5 for further information.

Please refer to the Tech Spec Punchlist item number in your response. Forward your response to G. Zinke.

J. C. Roberts
Chairman

LLJ/JCR:swb
cc: Mr. C. L. Tyrone
Mr. J. E. Cross
Mr. D. Stonestreet
Mr. A. S. McCurdy
Mr. S. Hutchins
Mr. J. Hendry
File (Tech Spec Records)

A4/61swb1

TO: M. FARSCHON (GE)

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 219 Priority: ✓1A

Identified By /Date Responsible Supervisor

Tech Spec Reference: Figure 3.4.6.1-1

Problem Title: Pressure/Temperature Limit Curves

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):

The pressure-temperature limit curves do not comply with the closure flange pressure-temperature safety margins in Program IV.A.2 of App. G to 10CFR50. Proof and review comment from MEB, December 15, 1983.

2. Safety Significance:

None, Latest App G change related to PWR concern rather than BWR.

3. Anticipated Resolution:

Evaluate to determine if the figure will require revision.

Revision needed to comply with latest App G (July 26, 1983)
Due to non safety concern (BWR) this issue can be deferred to later

4. NRC Response to Item (NRR/IE):

NRC Notified: Individual Notified Date / Time

5. Disposition:

Items Closed: (How)

Date / Time

c: J. L. C...
R. L. Pa...

"TECH SPEC PRIORITY"

Punchlist Item # See ATTAched

Tech Spec See ATTAched

Priority See ATTAched

TO: Manager of Nuclear Plant Engineering

FROM: Chairman, Prioritization and Disposition Chairman

SUBJECT: Technical Specifications Punchlist Item # See ATTAched

PDTS:84/ 0014

DATE: 3/10/84

The subject Tech Spec item has been determined by the Disposition Committee to require Engineering support.

DETAILS: This letter identifies requested response dates for the following Tech Spec problems:

# 199 Letter No. PDTS 84/0001	# 015 Letter No. PDTS 84/0007
# 180 Letter NO. PDTS 84/0002	# 198 Letter No. PDTS 84/0008
# 033 Letter NO. PDTS 84/0003	# 202 Letter NO. PDTS 84/0009
# 054 Letter NO. PDTS 84/0004	# 213 Letter NO. PDTS 84/0010
# 001 Letter NO. PDTS 84/0005	# 219 Letter No. PDTS 84/0011
# 016 Letter NO. PDTS 84/0006	# 118 Letter No. PDTS 84/0012

It is requested that the responses to the above items be completed by 3/13/84

Please contact Jerry RoberTs at Extension 2695 for further information.

Please refer to the Tech Spec Punchlist item number in your response. Forward your response to George ZINKE.

J. C. Roberts
Chairman

LLJ/JCR:swb

cc: Mr. C. L. Tyrone
Mr. J. E. Cross
Mr. D. Stonestreet
Mr. A. S. McCurdy
Mr. S. Hutchins
Mr. J. Hendry
File (Tech Spec Records)

A4/61swb1

Tech Spec Problem No.	Tech Spec	Priority
199	Table 3.3.6-1.5	1B
180	4.8.4.3	1D
033	Table 3.3.8-2	1B
054	3/4.3.8	1B
001	3/4.5.1	1B
016	3/4.3.8	1B
015	3/4.3.2	1D
198	3/4.3.7	1B
202	3/4.3.7	1B
213	3/4.3.3	1B
219	Figure 3.4.6.1-1	1B
168	3.6.3.1	1B

"TECH SPEC PRIORITY"

MEMO TO: J. F. Pinto, Manager of Nuclear Plant Engineering

FROM: C. L. Tyrone, Project Manager

SUBJECT: Handling of Tech Spec Review Items

TSRO: 84/0001

DATE: March 11, 1984

This memorandum confirms our conversation of March 10, 1984. At that time, your assistance was requested in resolving discrepancies on eleven priority 1 items. Since then two items have been added. These items are all previously identified items which require early resolution with the NRC. A response is needed on these items by 12:30 PM on March 11, 1984. A list is attached.

Furthermore, all items of any priority identified (or previously known) which are being handled on this program require expeditious handling. This includes areas where requests are originated from other interfacing organizations such as the Plant or Nuclear Services. In any case where conflicts regarding highest priority is not clear, I am available to provide clarification.

It is suggested that you arrange 7 day a week support in this area as it is needed and arrange for all NPE personnel who will be involved in this effort to be available (or on call) in a manner that will support the Tech Spec Review program.

CLT

SHH:sad
Attachment

cc: J. B. Richard (w/a
J. P. McGaughy (w/a)
J. F. Pinto (w/a)
J. E. Cross (w/a)
T. H. Cloninger (w/a)
H. J. Green (w/a)
R. C. Fron (w/a)
D. W. Stonestreet (w/a)
[redacted]
T. E. Reaves, Jr. (w/a)
S. M. Feith (w/a)
J. G. Cesare (w/a)
G. W. Smith (w/a)
L. R. McKay (w/a)
L. C. Burgess (w/a)
File (Tech Spec Records) (w/a)

LIST OF CURRENT PRIORITY 1
ITEMS REQUIRING NPE SUPPORT

PDTS:84/	P/L #	Date Sent
001	199	3/10/84
002	180	3/10/84
003	033	3/10/84
004	054	3/10/84
005	001	3/10/84
006	016	3/10/84
007	015	3/10/84
008	198	3/10/84
009	202	3/10/84
010	213	3/10/84
011	219	3/10/84
012	083	3/10/84
013	168	3/10/84

ENCLOSURE 7

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

219

DEC 15 1983

Docket No. 50-416

MEMORANDUM FOR: Cecil O. Thomas, Chief
Standardization & Special Projects Branch
Division of Licensing

FROM: B. D. Liaw, Chief
Materials Engineering Branch
Division of Engineering

SUBJECT: PROOF AND REVIEW OF GRAND GULF UNIT 1
TECHNICAL SPECIFICATIONS

As requested by your letter dated October 4, 1983, the Materials Engineering Branch, Division of Engineering, has reviewed the Grand Gulf Unit 1 Technical Specifications. The sections reviewed were 4.0.5, B4.0.5, and 3/4.4.6 on pages 3/4.4-17 thru 3/4.4-20. These sections are acceptable, except that the pressure-temperature limit curves in Figure 3.4.6.1-1 do not comply with the closure flange pressure temperature safety margins in Paragraph IV.A.2 of Appendix G, 10 CFR 50, which became effective on July 26, 1983. We will be sending to all licensees/applicants a generic letter informing them of the revised regulatory requirements. In accordance with that letter, the pressure-temperature limit curves in Figure 3.4.6.1-1 may require revision.

resp

This comment has been coordinated with Mr. Donald R. Hoffman of your staff.

B. D. Liaw, Chief
Materials Engineering Branch
Division of Engineering

cc: R. Vollmer
D. Eisenhut
F. Miraglia
W. Johnston
E. Sullivan
S. Pawlicki
D. Brinkman
B. Elliot
D. Hoffman
M. Houston
B. D. Liaw
C. Cheng
W. Hazelton
R. Klecker
M. Hum

Contact: B. Elliot
X-27741

~~8312230321~~

TO: M. FARSCHON (GE)

TECHNICAL SPECIFICATION PROBLEM SHEET

Item Number: 219 Priority: I A

Identified By / Date Responsible Supervisor

Tech Spec Reference: Figure 3.4.6.1-1

Problem Title: Pressure/Temperature Limit Curves

1. Problem Description (Tech Spec, FSAR, SER, GE Design, Other):

The pressure-temperature limit curves do not comply with the closure flange pressure-temperature safety margins in Program IV.A.2 of App. G to 10CFR50. Proof and review comment from MEB, December 15, 1983.

2. Safety Significance:

None, Latest App G change related to PWR concern rather than BWR.

3. Anticipated Resolution:

Evaluate to determine if the figure will require revision.

Revision needed to comply with latest App G (July 26, 1983)

Due to non safety concern (BWR) this issue can be deferred to later

4. NRC Response to Item (NRR/IE): ____

NRC Notified: ____ Individual Notified ____ Date / Time

5. Disposition: ____

Items Closed: (How) ____

____ / ____ Date Time

c: J. L. Cro...
R. L. Bo...

ENCLOSURE 2

Enclosure 2

Information Received from Licensee

Attachment 1 - Priority of potential Technical Specification changes

Attachment 2 - Sources of potential Technical Specification changes

Attachment 3 - Technical Specification problem sheets for 240 potential changes

Attachment 4 - Backup information for 32 high priority potential changes

Attachment 5 - January 10, 1983, letter from licensee to J. P. O'Reilly (NRC) and January 13, 1983, letter from R. C. Lewis (NRC) to licensee regarding interpretation of Technical Specifications 3/4 3.6 (Source Range Monitors)

ATTACHMENT 1

TECH SPEC PROBLEM SHEET PRIORITY DEFINITIONS

1. Problems Needing Resolution - Short Term

0 A. Safety Significant Item which would require plant shutdown, prohibit plant startup, or require other plant actions to reestablish safe operating conditions.

12 *B. Existing Tech Spec is non-conservative with respect to FSAR or supporting documents (e.g. approved design specs, SER, etc.). MP&L requires NRC concurrence and/or resolution prior to next criticality.

15 *C. Existing Tech Spec is non-conservative with respect to FSAR or supporting documents (e.g. approved design specs, SER, etc.). MP&L requires NRC concurrence and/or resolution prior to exceeding 5% Thermal Power.

2. Problems/Enhancements Needing Resolution - Long Term

4 not considered 1B, 3 considered 1C A. Existing condition could result in unnecessary challenges to safety systems or plant transients or is required to enhance plant safety.

38 B. Errors or confusing items in Technical Specifications which will not result in non-conservative operation with a reasonable dependence on administrative controls/plant knowledge/operational practices; Licensing commitments which require a Tech Spec change; items determined by MP&L to be important.

5 C. Could restrict power level or mode changes

76 D. Typographical Errors and Enhancements/Concerns which do not fall into a higher priority

26 E. Problems with, or enhancements to Tech Spec sections other than 3/4 (e.g. Administrative Controls, Bases, etc.)

7 F. Over-conservative Tech Specs for which changes are cost-justified

* The factors used to distinguish priorities 1B and 1C are operational mode requirements (generally Mode 2 requirements are associated with priority 1B and Mode 1 with priority 1C), fission product inventory considerations (generally priority 1B do not involve dealing with high fission product inventories while priority 1C which is associated with higher power levels may involve dealing with high fission produce inventories), and relative safety significance of systems.

Rev 2, 3/12/84

A2sd1

11 G. Design Changes which require Tech Spec changes

9 H. Pending design/analysis (e.g. Maximum Extended Operating Domain, Exxon Fuel, Single Recirc Loop Operation, etc.)

0 I. Others

3. Tech Spec change not justified (response required)

12 A. Item is generic and not included in STS

25 B. Others

2 Duplicates closed
6 issued in Amendment 12

249

Rev 2, 3/12/84

A2sd2

Rec'd 3/12/84 10:50pm

Best copy available

ATTACHMENT 2

INFORMATION FOR 3/9/84 NRC MEETING

Sources of Tech Spec Problem Items		# of Punchlist Items
1. Items identified by MP&L at the 1/24/84 meeting with NRC		61
2. NRC 2nd Proof and Review Comments given to MP&L informally ~~1/24/84~~ in onsite meeting with Hoffman 1/24/84		37
3. Items formally submitted to NRC prior to 1/24/84 (received 6 in Amend. 12)		43
	SUBTOTAL	141 end of 1/24/84 mtg
4. Identified by NRR informally		1
5. NRC I&E Exit (2/24/84) I&E Inspection 3/14/84		11
6. Additional Proof and Review Comments not previously punchlisted (items from more detail MP&L review of item 2 above)		11
7. PSRC Plant Safety Review Committee review of 141 subtotal above.		2
8. MP&L Review Team established to review LCO's/ACTION's (≈2/15/84)		39
9. QA MP&L Quality Assurance question.		1
10. LCTS (SER, Letters, FSAR, ~~etc.~~) Licensing commitment tracking system (LCTS) references to Tech Specs in SER, Letters, and FSAR		10
11. Instrumentation Review per ICSB Commitment Review Instrumentation for logic Tech Spec Instrumentation for consistent MOC in conjunction with NRC staff review.		10
12. Miscellaneous Technical Support identified items (primarily long-term issues and commitments) commitments for updates to Tech Spec due to startup Test results, Future changes for Exxon Fuel, MEOD analysis, (Maximum Extended Operational Domain)		14
	SUBTOTAL	99
	TOTAL	240 as of 3/9/84

13. Results of recent review efforts internal while present Tech Spec review effort was being established. 9

249

(2/11/84 memo)